# RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**APR1400 Design Certification**

**Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD**

**Docket No. 52-046**

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| **RAI No.:** | **461-8579** |
| **SRP Section:** | **15.06.03 - Radiological Consequences of Steam Generator Tube Failure (PWR)** |
| **Application Section:** | **15.06.03** |
| **Date of RAI Issue:** | **04/19/2016** |

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## Question No. 15.06.03-5

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, General Design Criteria (GDC) 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in Standard Review Plan (SRP) 15.0.3. NRC staff needs to ensure that a suitably conservative estimate is determined for the radiological release associated with the steam generator tube rupture event (SGTR).

NRC staff issued RAI 370-8450, Question No. 15.06.03-2, during the review of Section 15.6.3 of the APR1400 Design Control Document (DCD). In Question No. 15.06.03-2, NRC staff questioned the termination criteria for the analysis of the SGTR event. Steam relief through the main steam safety valves (MSSVs) on the affected steam generator is a significant contribution to the radiological consequences associated with the SGTR event. Chapter 15 safety analysis of the SGTR event typically extends to the point that the steam relief through the affected steam generator is terminated (i.e., steam relief through affected steam generator is terminated, break flow is terminated; see examples in AP1000 DCD, Palo Verde final safety analysis report (FSAR), McGuire FSAR, Catawba FSAR, Oconee FSAR). The analysis presented in the APR1400 DCD, however, is terminated once operator action is initiated at 30 minutes.

The KHNP response to RAI 370-8450, Question 15.06.03-2 did not alleviate NRC staff concerns because:

1. The RAI response failed to demonstrate that steam relief through the MSSVs on the

affected steam generator would be terminated with the initiation of operator action.

2. The RAI response contained information that appears to contradict information contained in the APR1400 DCD. Particularly, the RAI response states, "The analysis in the DCD conservatively assumed that the break flow rate at 30 minutes is maintained until the time at which the primary and secondary pressure are same, or the break flow is terminated." However, Table 15.6.3-5 of the APR1400 DCD shows that steam mass relief from the affected steam generator is 0.0 beyond 30 minutes (i.e. steam relief through the affected steam generator is terminated immediately upon initiation of operator action).

NRC staff requests that KHNP:

1. Extend the CESEC-III analysis of the SGTR event until acceptable analysis termination criteria are met (e.g. steam relief through affected steam generator is terminated, break flow is terminated)

2. Update Section 15.6.3 of the APR1400 DCD as appropriate.

3. Update the analysis of radiological consequences as appropriate.

## Response

1. Steam relief through the MSSVs on the affected steam generator would be terminated with the isolation of affected SG as shown in DCD Table 15.6.3-3 (once operator action is initiated at 30 minutes) and DCD Figure 15.6.3-23 show the relevant results. After 30 minutes, steam mass release from the unaffected SG is assumed through the atmospheric dump valve (ADV) as shown in DCD Table 15.6.3-5. Only the break flow is maintained until the primary and secondary pressures are same. DCD Table 15.6.3-5 separately shows steam mass relief from the affected steam generator before the isolation of the affected SG, steam mass relief from the unaffected SG, and break flow until the time at which the primary and secondary pressure are same.

   As shown in the Figure 1 below, the primary and secondary pressures are same at 4,000 seconds after the event initiation. The break flow is decreased after 1,800 seconds and terminated at 4,000 seconds as shown in Figure 2. However, in terms of radiological consequences, the results in DCD conservatively assumed that the break flow rate at 1,800 seconds is maintained until that the primary and secondary pressures are equalized in the calculation of the flashed mass of break flow. The total flashed mass of break flow is assumed to be completely discharged to the environment. Therefore, DCD analysis presents a bounding case in term s of radiological consequences.

   Since the results from the assumption applied in DCD are conservative and the operator action is not credited until 30 minutes after event initiation event, extended CESEC-III results to show the time at which the primary and secondary pressure are same and the break flow is terminated are not presented.

2. There is no update to Section 15.6.3 of the APR1400 DCD based on Response 1 evaluation provided above. The case analyzed for the DCD already considers the extended CESEC-III results in terms of radiological consequences as shown in Figure 2 showing the

time at which the primary and secondary pressure are same and the break flow is terminated.

3. There is no update to the analysis of radiological consequences based on Response 1 evaluation provided above. The case in DCD already presents the limiting case in terms of radiological consequences considering the extended CESEC-III results in terms of break flow termination.

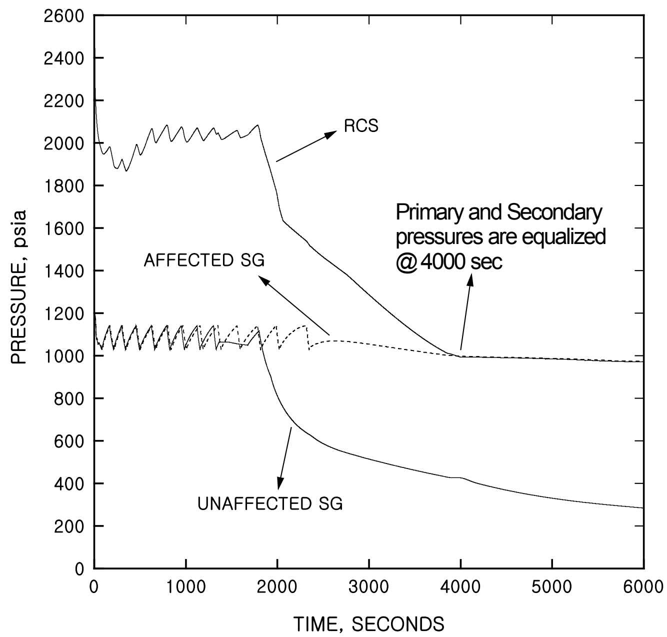


**Figure 1 Primary and Secondary Pressures vs. Time**

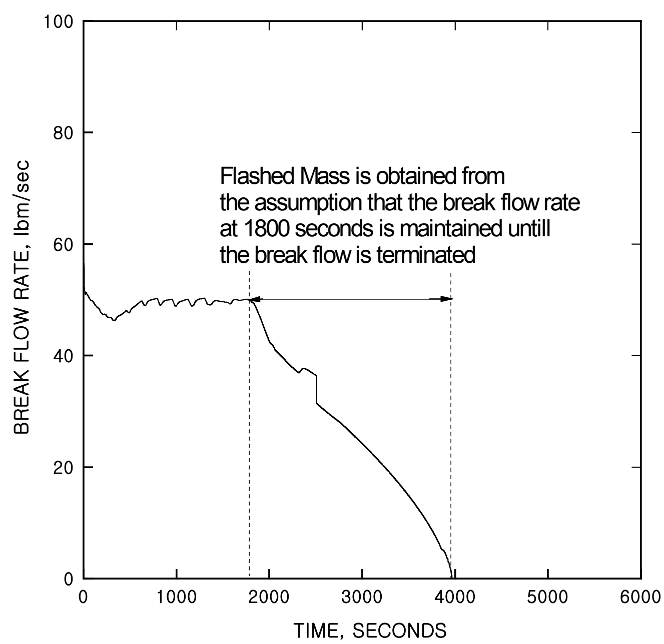


**Figure 2 Tube Leak Rate vs. Time**

**Impact on DCD**

There is no impact on the DCD.

**Impact on PRA**

There is no impact on the PRA.

**Impact on Technical Specifications**

There is no impact on the Technical Specifications.

**Impact on Technical/Topical/Environmental Reports**

There is no impact on any Technical, Topical, or Environment Report.